UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 11, 2019

Mr. Ernest J. Kapopoulos, Jr.
Site Vice President
H. B. Robinson Steam Electric Plant
Duke Energy Progress, LLC
3581 West Entrance Road, RNPA01
Hartsville, SC 29550

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – STAFF EVALUATION RELATED TO THE 2017 ANNUAL REPORT OF CHANGES AND ERROR CORRECTIONS AFFECTING THE LARGE-BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS (EPID L-2018-LRO-0028)

Dear Mr. Kapopoulos:

By letter dated May 24, 2018, Duke Energy (the licensee), submitted its 2017 annual report of changes and error corrections affecting the evaluation models used to demonstrate acceptable emergency core cooling system performance for its licensed facilities (Reference 1). Duke Energy identified no new changes or errors affecting the calculated peak cladding temperature for the large-break loss-of-coolant accident analysis for the H. B. Robinson Steam Electric Plant, Unit 2.

This report was submitted pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 46 (10 CFR 50.46), paragraph (a)(3).

The U.S. Nuclear Regulatory Commission (NRC) staff has evaluated the report and, while performing other regulatory activities, were made aware of an error affecting the evaluation model used to demonstrate acceptable emergency core cooling system performance for H. B. Robinson for the large-break loss-of-coolant accident. The error, which was identified in 2017 by the vendor for the evaluation model (Framatome), involved the neglect of cladding deformation in the calculation of the metal-water reaction (i.e., cladding oxidation) and appeared to have the potential to affect the calculated peak cladding temperature and other figures of merit.

As a result, the NRC staff, on October 17, 2018, issued a request for additional information (RAI) to facilitate completion of the review (Reference 2). In a letter dated December 10, 2018, Duke Energy provided a response to the RAI (Reference 3).

Based on its evaluation, the NRC staff determined that the report, including the licensee's responses, satisfies the reporting requirements of 10 CFR 50.46(a)(3). A staff evaluation describing the technical and regulatory basis for the NRC staff's conclusion is enclosed. This letter concludes the NRC staff's review associated with EPID L-2018-LRO-0028.

If you have any questions, please contact me at (301) 415-7410 or via e-mail at Natreon.Jordan@nrc.gov.

Sincerely,

Natreon Jordan, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-260

Enclosure:
Staff Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

STAFF EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO THE 2017 ANNUAL REPORT OF CHANGES AND ERROR CORRECTIONS AFFECTING THE LARGE-BREAK LOSS-OF-COOLANT-ACCIDENT ANALYSIS PURSUANT TO 10 CFR 50.46 FOR THE H. B. ROBINSON NUCLEAR POWER PLANT, UNIT 2

1.0 INTRODUCTION

Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.46(a)(3)(ii), on May 24, 2018, Duke Energy submitted an annual report of changes and error corrections affecting the evaluation models used to demonstrate acceptable emergency core cooling system (ECCS) performance for its licensed facilities (Reference 1). Duke Energy stated, therein, that no changes or errors affecting the calculated peak cladding temperature for the large-break loss-of-coolant accident (LOCA) analysis for the H. B. Robinson Steam Electric Plant, Unit 2, were identified during the 2017 reporting period.

However, in the course of performing other regulatory activities, the NRC staff learned of an error affecting the evaluation model used by Duke Energy to demonstrate acceptable ECCS performance for H. B. Robinson for the large-break LOCA event. According to Framatome, the vendor for the evaluation model, the error involved the neglect of cladding deformation in the calculation of the metal-water reaction (i.e., cladding oxidation). They further specified that the error was identified by Framatome in 2017. The U.S. Nuclear Regulatory Commission (NRC or the Commission) staff determined that the error could have the potential to affect the peak cladding temperature and other figures of merit calculated for H. B. Robinson for the large-break LOCA event.

Consequently, the NRC staff initiated a review of Duke Energy's 2017 annual report of changes and error corrections for H. B. Robinson, and, on October 17, 2018, issued a request for additional information (RAI) to facilitate completion of the review (Reference 2). Duke Energy provided responses to the RAIs via a letter dated December 10, 2018 (Reference 3).

2.0 REGULATORY EVALUATION

2.1 Regulatory Requirements

Regulatory requirements applicable to ECCS performance in the event of a postulated LOCA are provided, in part, by 10 CFR 50.46. Among these requirements, 10 CFR 50.46 specifies certain standards for acceptable evaluation models, acceptance criteria for ECCS performance, and reporting requirements associated with changes and error corrections to acceptable evaluation models.

In particular, 10 CFR 50.46(a)(3)(ii) specifies that

> For each change to or error discovered in an acceptable evaluation model or in the application of such a model that affects the temperature calculation, the applicant or holder of a construction permit, operating license, combined license, or manufacturing license shall report the nature of the change or error and its estimated effect on the limiting ECCS analysis to the Commission at least annually.

Further, 10 CFR 50.46(a)(iii) specifies, in part, that

> If the change or error is significant, the applicant or licensee shall provide this report within 30 days and include with the report a proposed schedule for providing a reanalysis or taking other action as may be needed to show compliance with § 50.46 requirements.

A significant change or error is defined by 10 CFR 50.46(a)(3)(i) as involving a difference of 50 degrees Fahrenheit (°F) or more in the calculated peak cladding temperature, relative to the value previously calculated using the last acceptable model.

Furthermore, if a change or error correction results in calculated ECCS performance that does not comply with the acceptance criteria specified in 10 CFR 50.46(b), then 10 CFR 50.46(a)(3)(ii) further requires that the affected licensee propose immediate steps either to demonstrate compliance with 10 CFR 50.46(b) or to bring the plant design or operation into compliance therewith.

The three quantitative acceptance criteria specified in 10 CFR 50.46(b) are summarized below in Table 1.

Table 1: 10 CFR 50.46(b) Quantitative Acceptance Criteria

Figure of Merit	Acceptance Criterion
Peak Cladding Temperature	≤ 2200 °F
Maximum (Local) Cladding Oxidation	≤ 17% of Unoxidized Thickness
Maximum (Core-Wide) Hydrogen Generation	≤ 1% of Hypothetical Amount

Additionally, 10 CFR 50.46(b) contains two qualitative acceptance criteria, namely that (1) the reactor core shall be maintained in a coolable geometry, and (2) adequate long-term core cooling shall be provided to remove decay heat for the period required by the long-lived radioactivity remaining in the reactor core.

3.0 BACKGROUND

3.1 Current Evaluation Model

The large-break LOCA analysis for H. B. Robinson is performed in accordance with the Realistic Large Break LOCA evaluation model developed by Framatome. The Realistic Large Break

LOCA methodology uses the S-RELAP5 thermal-hydraulic code to perform a best-estimate analysis of ECCS performance that explicitly accounts for calculational uncertainties.

Framatome submitted the original version of this evaluation model, as described in topical report EMF-2103, Revision 0, for NRC staff review on August 20, 2001 (Reference 4). The NRC staff approved the methodology on April 9, 2003 (Reference 5). Subsequently, in an amendment request dated March 3, 2005, the licensee requested approval for H. B. Robinson to adopt Revision 0 of the Realistic Large Break LOCA methodology (Reference 6). The NRC staff approved the request in a safety evaluation dated September 20, 2006 (Reference 7).

Duke Energy's 2017 annual report of changes and error corrections (Reference 1) identifies that the licensing-basis peak cladding temperature for the large-break LOCA event for H. B. Robinson, calculated in accordance with its existing evaluation model, is 2088 °F.

3.2 Evaluation Model Error

Prior to Framatome's recognition of the error in 2017, S-RELAP5 had not been programmed to correctly model the impact of cladding deformation on the calculation of cladding oxidation. In particular, a LOCA event may result in both high fuel rod cladding temperatures and low reactor coolant system pressures. Under such conditions, some fraction of the fuel rods in the core may experience localized cladding swelling (i.e., increase in diameter) and rupture.

Neglecting the influence of such cladding deformation in the calculation of cladding oxidation had the potential to exert a nonconservative influence on predictions of the figures of merit shown above in Table 1 for reasons including the following:

- As fuel rod cladding swells, its wall thickness necessarily decreases (i.e., in accordance with the law of conservation of mass), thereby reducing the margin available to the acceptance criterion for local cladding oxidation. Neglecting this effect would be inconsistent with the explicit requirement in 10 CFR 50.46(b)(2) that thinning of the cladding wall due to swelling must be considered in this determination.

- As fuel rod cladding swells, its surface area increases. Neglecting the impact of the expanded surface area would result in underestimation of the exothermic metal-water reaction rate.

- When fuel rod cladding ruptures, the cladding adjacent to the rupture location would be susceptible to the exothermic metal-water reaction on both its outer and inner surfaces.

Neglecting cladding deformation may also give rise to certain conservative impacts, which include overestimation of the pellet-to-cladding heat transfer coefficient and underestimation of the cladding surface area available for transferring heat to the coolant. However, as the cladding temperature increases, due to the exponential dependence of the metal-water reaction rate upon temperature, the NRC staff generally expects the neglect of cladding deformation to exert an overall nonconservative influence on the predicted figures of merit used to demonstrate compliance with the acceptance criteria in 10 CFR 50.46(b).

4.0 EVALUATION

4.1 Impact of S-RELAP5 Code Error on EMF-2103, Revision 0

The NRC staff's approval of EMF-2103, Revision 0, did not require the Realistic Large Break LOCA evaluation model to account for cladding swelling and rupture explicitly (References 4, 5).[1] As such, some reactor licensees referencing Revision 0 of the Realistic Large Break LOCA evaluation model may have anticipated no impact from the S-RELAP5 code error associated with the neglect of cladding deformation in the computation of cladding oxidation.

However, the NRC staff's review of the issue identified that:

- The origin of the S-RELAP5 code error predates Framatome's 2001 submittal of the Realistic Large Break LOCA evaluation model.

- Sensitivity calculations performed with a version of the S-RELAP5 code containing the modeling error served as the primary basis for the NRC staff's decision to approve EMF-2103, Revision 0, without an explicit accounting for cladding swelling and rupture.

Expanding upon the latter bullet, the NRC staff reviewed Appendix B of EMF-2103, Revision 0 (Reference 4), which identifies perceived conservatisms in the Realistic Large Break LOCA evaluation model. In this appendix, Framatome emphasized the primacy of sensitivity calculations performed with the version of S-RELAP5 containing the modeling error by stating, in part:

> Among the major assumptions stated for the FRA-ANP RLBLOCA [Framatome-ANP Realistic Large Break LOCA] methodology are declarations of adopted conservatism. Such declarations are not always physically intuitive. In these instances, sensitivity studies have been performed to arrive at the stated conclusions. In this appendix, selections of calculations are presented to support some of the statements of conservatism presented in this methodology document.

Appendix B to EMF-2103, Revision 0, cites four specific conservatisms Framatome perceived in the Realistic Large Break LOCA evaluation model, one of which is discussed in Section B.2, "Analysis without Clad Swelling and Rupture." Section B.2 describes and documents the results of sensitivity studies that Framatome performed using the S-RELAP5 code to reach the conclusion that it would be conservative for the Realistic Large Break LOCA evaluation model to neglect cladding swelling and rupture.

In its responses to RAIs 28, 96, and 132 from the NRC staff's review of EMF-2103, Revision 0 (Reference 8), Framatome provided additional context and support for its position that neglecting cladding swelling and rupture would be conservative:

---

[1] Note, however, that certain license amendment requests to implement EMF-2103, Revision 0, that followed H. B. Robinson were approved only after Framatome augmented the Realistic Large Break LOCA evaluation model on a plant-specific basis to explicitly model fuel rod cladding swelling, cladding rupture, and fuel relocation (e.g., Shearon Harris Nuclear Plant (Reference 10)).

- Framatome's response to RAI 28 states, in part that:

  Swelling and rupture models were not used in the Framatome methodology because use of the swelling and rupture models based on NUREG-0630 would yield slightly reduced PCTs [peak cladding temperatures]....

- Framatome's response to RAI 96 cites the sensitivity studies performed in Section B.2 of Appendix B to EMF-2103, Revision 0, as the basis for characterizing the general influence of fuel rod swelling and rupture as "relatively small and beneficial."

- Framatome's response to RAI 132 discusses an additional sensitivity study performed using the S-RELAP5 code that appears to show that neglect of swelling and rupture is conservative even in a case where rupture of the fuel rod cladding occurs.

However, as noted above, the version of the S-RELAP5 code used to perform all of the sensitivity calculations contained in EMF-2013, Revision 0, and associated RAI responses was affected by the error described above associated with the neglect of cladding deformation in the calculation of cladding oxidation. Hence, the NRC staff had no basis for continued confidence in the results of the sensitivity studies and derivative conclusions described by Framatome in Section B.2 of Appendix B to EMF-2103, Revision 0, and associated RAI responses. Furthermore, the NRC staff discovered recent estimates performed for other affected pressurized-water reactors (e.g., Sequoyah Nuclear Plant, Units 1 and 2 (Reference 9), and Shearon Harris Nuclear Power Plant, Unit 1 (Reference 1)) that are intended to correct for the influence of the S-RELAP5 modeling error. These recent estimates indicate that, contrary to the information submitted by Framatome in support of the NRC staff's review of EMF-2103, Revision 0, the neglect of swelling and rupture in the Realistic Large Break LOCA methodology is actually (1) nonconservative and (2) potentially significant in magnitude (i.e., greater than 50 °F).

Based upon the review summarized above, the NRC staff determined that the S-RELAP5 code error concerning the neglect of cladding swelling and rupture on the calculation of cladding oxidation had apparently led to a further erroneous conclusion in EMF-2103, Revision 0, that the neglect of cladding swelling and rupture is conservative relative to explicit modeling of these phenomena. As a result of this, the NRC staff requested additional information from Duke Energy to characterize the impact of correcting these modeling errors on the large-break LOCA analysis for H. B. Robinson.

4.2 <u>Request for Additional Information</u>

On October 17, 2018, the NRC staff requested that Duke Energy provide additional information concerning the impact of the errors associated with the neglect of cladding swelling and rupture on the calculation of cladding oxidation in the H. B. Robinson large-break LOCA analysis (Reference 2). More specifically, the NRC staff requested that the licensee provide the following information:

(a) a revision to the 2017 annual report of changes and errors submitted pursuant to 10 CFR 50.46(a)(3)(ii) that acknowledges and estimates the impact of the modeling errors in the existing large-break LOCA evaluation model applied to H. B. Robinson that are associated with (1) the incorrect computation of cladding oxidation and (2) the nonconservative neglect of cladding swelling and rupture based upon the vendor's submission of erroneous information,

(b) clarification as to whether the impact of the modeling errors was significant (i.e., having an effect greater than 50 °F on the calculated peak cladding temperature and to further provide a 30-day error report in accordance with 10 CFR 50.46(a)(3)(ii),

(c) confirmation that all acceptance criteria of 10 CFR 50.46(b) remain satisfied once the effect of the modeling errors described above has been taken into account, or a description of immediate steps necessary to bring plant design or operation into compliance in accordance with 10 CFR 50.46(a)(3)(ii), and

(d) adequate description of and justification for the method used to estimate the impacts of the errors described above.

4.3 Licensee's RAI Response

On December 10, 2018, Duke Energy provided additional information in response to the NRC staff's request (Reference 3), which is summarized as follows:

(a) Duke Energy estimated that correction of the modeling errors described above would result in an increase of 31 °F in the calculated peak cladding temperature for the large-break LOCA event, resulting in a licensing-basis peak cladding temperature value of 2119 °F for H. B. Robinson. Duke Energy's response included a revision to the 2017 annual report of changes and error corrections that acknowledged the errors in Revision 0 of the Realistic Large Break LOCA evaluation model described above and identified their impact on the calculated peak cladding temperature.

(b) Duke Energy's estimate of the impact of the modeling errors on the calculated peak cladding temperature for the large-break LOCA event did not exceed the 50 °F significance threshold defined in 10 CFR 50.46(a)(3)(i).

(c) Duke Energy concluded that the acceptance criteria in 10 CFR 50.46(b) for the peak cladding temperature and other figures of merit remain satisfied for the large-break LOCA event once the modeling errors described above have been corrected.

(d) Duke Energy stated that the estimated impact of the modeling errors was determined via explicit analyses. More specifically, Duke Energy performed the estimate using a version of the S-RELAP5 code that is identical to that used for the previous analysis of record, excepting the modifications necessary to correct the errors associated with the modeling of cladding deformation and oxidation. The estimate further used the input file from the case in the existing analysis of record that set the limiting value for peak cladding temperature.

4.4 Evaluation of Licensee's RAI Response

The NRC staff determined that the additional information submitted by Duke Energy is sufficient to address the issues identified during the staff's review of the 2017 annual report of changes and error corrections affecting the evaluation model used to demonstrate acceptable ECCS performance for the large-break LOCA event for H. B. Robinson. The NRC staff's determination is based upon the following factors:

- Duke Energy submitted a revised annual report of changes and errors for the 2017 reporting period on December 10, 2018. The revised annual report acknowledged the modeling errors described above and estimated their impact, as required by 10 CFR 50.46(a)(3)(ii).

- Duke Energy confirmed that the impact of the errors was not significant, as defined in 10 CFR 50.46(a)(3)(i).

- Duke Energy confirmed that all acceptance criteria in 10 CFR 50.46(b) remain satisfied.

- Use of an explicit analysis with a corrected version of the existing evaluation model in lieu of more simplified techniques provides increased assurance in the accuracy of the estimated impact. Furthermore, basing the estimate upon the case from the analysis of record with the highest peak cladding temperature is appropriate based on the following:

    - peak cladding temperature is the figure of merit in the H. B. Robinson large-break LOCA analysis with the least margin to the acceptance criteria in 10 CFR 50.46(b),

    - the reporting of errors and changes under the existing regulation (i.e., 10 CFR 50.46) is focused primarily upon the effect on peak cladding temperature,

    - the case in question underwent both cladding swelling and rupture, which resulted in the oxidation of both the outer and inner cladding surfaces at the rupture node, and

    - the rate of the metal-water reaction on the cladding has an exponential dependence upon the cladding temperature.

5.0 CONCLUSION

Based upon the review described above, the NRC staff determined that the revised annual report of changes and errors for the 2017 reporting period submitted for H. B. Robinson by Duke Energy on December 10, 2018 (Reference 3), satisfies the reporting requirements specified in 10 CFR 50.46(a)(3)(ii). The NRC staff further concluded that Duke Energy performed an adequate estimate of the impact of the modeling errors on the calculated figures of merit specified in 10 CFR 50.46(b). Hence, based upon the information provided by Duke Energy, there is reasonable assurance that H. B. Robinson continues to satisfy the acceptance criteria specified in 10 CFR 50.46(b).

6.0 REFERENCES

1. Donahue, J., Duke Energy, letter to U.S. Nuclear Regulatory Commission, "Carolinas LLC (Duke Energy) Annual Report of Changes Pursuant to 10 CFR 50.46," May 24, 2018 ((Agencywide Documents Access and Management System (ADAMS) Accession No. ML18150A705).

2. Galvin, D., U.S. Nuclear Regulatory Commission, email to A. Zaremba, Duke Energy, "Robinson RAIs – Duke Energy 10 CFR 50.46 Annual Report (EPID-L-2018-LRO-0028)," October 17, 2018 (ADAMS Accession No. ML18291A644).

3. Donahue, J., Duke Energy, letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information (RAI) Regarding 10 CFR 50.46 Annual Report, Including Revised Robinson Large Break Loss of Coolant Accident Report," December 10, 2018 (ADAMS Accession No. ML18344A656).

4. Mallay, J.F., Framatome ANP, letter to U.S. Nuclear Regulatory Commission "Request for Review of EMF-2103(P) Revision 0, Realistic Large Break LOCA Methodology for Pressurized Water Reactors," August 20, 2001 (ADAMS Accession No. ML012400042).

5. Berkow, H.N., U.S. Nuclear Regulatory Commission, letter to J.F. Mallay, Framatome ANP, "Safety Evaluation on Framatome ANP Topical Report EMF-2013(P), Revision 0, 'Realistic Large Break Loss-of-Coolant Accident Methodology for Pressurized Water Reactors'," April 9, 2003 (ADAMS Package Accession No. ML030760337).

6. Lucas, J.F., Progress Energy, "Request for Technical Specifications Change to Core Operating Limits Report (COLR) References," March 3, 2005 (ADAMS Package Accession No. ML050700404).

7. Mozafari, B.L., U.S. Nuclear Regulatory Commission, letter to T.D. Walt, Carolina Power & Light Company, "H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of Amendment Regarding Methodology for Large Break Loss-of-Coolant Accident Analyses (TAC No. MC6630), September 20, 2006 (ADAMS Package Accession No. ML062330026).

8. Mallay, J.F., Framatome ANP, letter to U.S. Nuclear Regulatory Commission "Publication of EMF-2103(P)(A) Revision 0, 'Realistic Large Break LOCA Methodology for Pressurized Water Reactors'," September 15, 2003 (ADAMS Package Accession No. ML032691410).

9. Williams, A.L., Tennessee Valley Authority, letter to U.S. Nuclear Regulatory Commission, "10 CFR 50.46 30-Day Report for Sequoyah Nuclear Plant, Units 1 and 2," January 18, 2018 (ADAMS Accession No. ML18018B158).

10. Billoch Colon, A.T., U.S. Nuclear Regulatory Commission, letter to C.L. Burton, Progress Energy Carolinas, Inc., "Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment Re: The Revision to Technical Specification Core Operating Limits Report References for Realistic Large Break Loss-of-Coolant Accident Analysis (TAC No. ME6999)," May 30, 2012 (ADAMS Accession No. ML12076A103).

Principle Contributor: John Lehning

Date: June 11, 2019

SUBJECT: H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – STAFF EVALUATION RELATED TO THE 2017 ANNUAL REPORT OF CHANGES AND ERROR CORRECTIONS AFFECTING THE LARGE-BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS (EPID L-2018-LRO-0028) DATED JUNE 11, 2019

DISTRIBUTION:
PUBLIC
RidsRgn2MailCenter Resource
RidsNrrDorlLPL2-2 Resource
RidsNrrPMRobinson Resource
RidsNrrLABClayton
RidsAcrsAcnw_MailCTR Resource
RidsNrrDssSnpb Resource
RLukes, NRR
UShoop, NRR
NJordan, NRR
JLehning, NRR
LPL2-2 R/F

ADAMS Accession No.: ML19121A482 *via memo dated

OFFICE	LPL2-2/PM	LPL2-2/LA	DSS/SNPB*	LPL2-2/BC	LPL2-2/PM
NAME	NJordan	BClayton	RLukes	UShoop	NJordan
DATE	5/25/19	5/30/19	3/14/2019	6/10/19	6/11/19

OFFICIAL RECORD COPY